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RADIATION TRANSPORT CALCULATIONS OF A SIMPLE STRUCTURE USING THE VEHICLE CODE SYSTEM WITH 69-GROUP CROSS SECTIONS AND THE MONTE-CARLO NEUTRON AND PHOTON CODE

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HOWELL CATON

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The 58-Group Cross-Sections (DLC-31) used in VCS calculations do not adequately describe secondary gamma production in steel. In response, Oak Ridge National Laboratory (ORNL) has compiled a set of Cross sections with 69 energy groups. This report documents an escalation of the 69 group cross sections on a simple steel box with various radiation liners. Also evaluated is the Monte Carlo Neutron and Photon Transport Code (MPNP) written at Los Alamos National Laboratory (LANL).

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I. INTRODUCTION

1. Objective

The objectives of this study are as follows:

a. Evaluate the 69-group cross-section set being developed by the Oak Ridge National Laboratory for the Defense Nuclear Agency as a replacement for the 58-group DLC-31 in the Vehicle Code System (VCS).

b. Evaluate the usefulness of the Monte Carlo Neutron and Photon Transport Code System (MCNP) for armored vehicle radiation shielding problems.

2. Background

The principle radiation transport code used for armored vehicle shielding problems is the Vehicle Code System (VCS) (described in section 3). Work over the past few years indicates that there are certain situations in which the cross-section detail used for VCS is inadequate. When thermal neutrons are absorbed in iron, gammas are produced which can add significantly to the radiation present. In some cases, VCS has estimated very low secondary gamma production. The approach taken to increase the level of cross-section detail was to switch to another cross-section set having 46 neutron groups and 23 gamma groups (69 groups total). The new set is explained more thoroughly in section 4.

The MCNP code, a transport code written at Los Alamos National Laboratory, has several advantages over the VCS methodology. MCNP has better documentation, less cryptic error messages, more flexible tally systems, mnemonic input cards, and far greater cross-section detail than does the 69 group cross-section set. Section 5 describes the MCNP code in greater detail.

The purpose of this report is to evaluate both the 69 group cross-sections and the MCNP code by using them on simple shielding configurations for which the 58 group cross- sections were inadequate. These calculations are compared to VCS calculations using the 58 group cross- sections and to measurements performed at the Army Pulsed Radiation facility. VCS calculations using 69-group cross-sections will be referred to as VCS/69. VCS with the 58-group set will be referred to simply as VCS.

3. The Vehicle Code System

The Vehicle Code System, VCS,[1,2] is a modular computer code employed in the solution of the problem of radiation transport from the weapon to a target some distance away. Because of the large separation distances usually considered, the problem can be restated as being essentially the transport of radiation from a point source to a point detector.

In VCS, the radiation transport problem is separated into two major calculations. A discrete ordinates solution of the Boltzmann Equation for the fluence in the vicinity of the target resulting from the weapon burst is obtained with the Discrete Ordinates Transport (DOT) computer code.[3] The free-field fluxes are recorded on a binary tape known as a VISA tape. A special version of the Monte Carlo radiation transport code, MORSE,[4] called MIFT,[5] performs the second calculation. MIFT operates in the adjoint mode to provide an importance function at a surface surrounding the target structure. In the adjoint mode, particles are traced "backwards" from detector to source. The MIFT code uses the GIFT-IV geometry code which has been incorporated into the MORSE code at BRL. The importance function is a measure of the probability that a particle existing at the surrounding surface, or its produced secondary particle, will reach the crew member in the vehicle. The fluence and the dose at the detector's position is obtained by a computer code called the Detector Response Code (DRC), which weights the free-field flux by the importance function.

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1. W.A. Rhoades, *Development of a Code System for Determining Radiation Protection of Armored Vehicles* , Oak Ridge National Laboratory, ORNL-TM-4664 (1974). (UNCLASSIFIED)

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3. F.R. Mynatt, F.J. Muckenthaler and P.N. Stevens, *Development of Two-Dimensional Discrete Ordinates Transport Theory for Radiation Shielding* , Union Carbide Corporation (Nuclear Division), CTC-INF-952 (1969). (UNCLASSIFIED)

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4. The 69-group Cross-Section Set

The 69-group cross-section set increases the level of cross-section detail by dividing the old groups into two or more groups, The upper and lower energy levels for the neutrons are retained. The total number of neutron groups is increased from 37 to 46. The upper limit of the gamma set is increased from 14 Mev to 20 Mev. The new energies make up one additional group. Another new gamma group is creating by dividing the highest energy group in the old set into two groups. This increases the number of gamma groups from 21 to 23. The increased detail is expected to improve the secondary gamma generation in iron. Tables 1 and 2 compare the energy boundaries of the two cross-section sets.

5. The Monte Carlo Neutron and Photon Transport Code (MCNP)

MCNP was written at Los Alamos National Laboratory. It is currently maintained by the X6 Group at that location. It is one of the oldest and most extensive monte carlo programs in the United States, being a direct descendant of original monte carlo work by Fermi, von Neumann, and Ulam. It is written in Fortran 77, and is largely machine independent. MCNP operates in the neutron only mode, the gamma only mode, and the coupled neutron-gamma mode. The gamma treatment permits the simple mode (Compton Scattering, Pair Production, and Photoelectric Effect) and the detailed treatment (includes Coherent/Incoherent Scattering, Florescent Emission). Problems may be stationary or time dependent, and fixed or fission sources are allowed. Although, MCNP works in the forward mode only (i.e. makes source to detector calculations), there is a multigroup patch (the MCMG Code) that has both forward and adjoint modes.

A wide variety of source conditions can be input without having to make code modifications. Independent probability distributions are allowed for energy, time, position, and direction; or source variables may depend upon other variables (example: direction as a function of energy). All input distributions can be biased, and the following built-in functions are available: fission and fusion energy spectra such as Watt, Maxwellian, and Gaussian spectra; Gaussian for time; and isotropic, cosine, and monodirectional for direction.

Cross-section files for MCNP are large and detailed. Cross-sections are tabulated for continuous energies with linear interpolation between specific energies. However, particles can be treated in the broad-group fashion as well. Cross-sections are reportedly accurate in most cases to 0.5 percent. Neutron data is from 20 MeV to .00001 eV, and gammas -- 100 Mev to 1 KeV. Discrete neutron X-sections are also available -- 262 Groups. Neutron X-sections are available for 80 Nuclides.

Table 1. Neutron Cross-section Sets

Group No.	Upper Energy Boundary (eV) 37 Groups	46 Groups
1	.19640e+08	.19640e+08
2	.16905e+08	.16905e+08
3	.14918e+08	.14918e+08
4	.14191e+08	.14191e+08
5	.13840e+08	.13840e+08
6	.12523e+08	.12523e+08
7	.12214e+08	.12214e+08
8	.11052e+08	.11052e+08
9	.10000e+08	.10000e+08
10	.90484e+07	.90484e+07
11	.81873e+07	.81873e+07
12	.74082e+07	.74082e+07
13	.63763e+07	.63763e+07
14	.49659e+07	.49659e+07
15	.47237e+07	.47237e+07
16	.40657e+07	.40657e+07
17	.30119e+07	.30119e+07
18	.23852e+07	.23852e+07
19	.23069e+07	.23069e+07
20	.18268e+07	.18268e+07
21		.14227e+07
22	.11080e+07	.11080e+07
23		.96164e+06
24		.82085e+06
25		.74274e+06
26		.63928e+06
27	.55023e+06	.55023e+06
28		.36883e+06
29		.24724e+06
30	.15764e+06	.15764e+06
31	.11109e+06	.11109e+06
32	.52475e+05	.52475e+05
33		.34307e+05
34	.24788e+05	.24788e+05
35	.21785e+05	.21785e+05
36	.10333e+05	.10333e+05
37	.33546e+04	.33546e+04
38	.12341e+04	.12341e+04
39	.58295e+03	.58295e+03
40		.27536e+03
41	.10130e+03	.10130e+03
42	.29023e+02	.29023e+02
43	.10677e+02	.10677e+02
44	.30590e+01	.30590e+01
45	.11250e+01	.11250e+01
46	.41400e+00	.41400e+00

Table 2. Gamma Cross-section Sets

Group No.	Upper Energy Boundary (eV) 21 Groups	23 Groups
1		.20000e+08
2	.14000e+08	.14000e+08
3		.12000e+08
4	.10000e+08	.10000e+08
5	.80000e+07	.80000e+07
6	.70000e+07	.70000e+07
7	.60000e+07	.60000e+07
8	.50000e+07	.50000e+07
9	.40000e+07	.40000e+07
10	.30000e+07	.30000e+07
11	.25000e+07	.25000e+07
12	.20000e+07	.20000e+07
13	.15000e+07	.15000e+07
14	.10000e+07	.10000e+07
15	.70000e+06	.70000e+06
16	.45000e+06	.45000e+06
17	.30000e+06	.30000e+06
18	.15000e+06	.15000e+06
19	.10000e+06	.10000e+06
20	.70000e+05	.70000e+05
21	.45000e+05	.45000e+05
22	.30000e+05	.30000e+05
23	.20000e+05	.20000e+05

The basic unit for MCNP geometry is the "cell", which is used to identify particular regions with particular materials, define importance regions for variance reduction, and to collect results (tallies). Cells are delimited by surfaces of first or second algebraic order and certain surfaces of fourth order (tori). Geometric input is very flexible.

6. The Army Pulsed Radiation Facility (APRF)

The Army Pulsed Radiation Facility, which simulates a fission weapon, is a bare critical assembly in the form of a right circular cylinder 22.6 cm in diameter and 19.8 cm in height. The reactor is mounted on a transporter and was positioned outdoors 14 ± 0.5 cm above the surrounding terrain for all the measurements on the box. The reactor simulates a fission nuclear weapon in the type of radiation it emits.

The reactor data are related to nominal kilowatt hours (kWhr) of reactor operation. The integrated leakage rate is 1.10×10^{17} (±15%) source neutrons per kWhr. There are approximately 0.65 (±15%) source gamma ray per source neutron.

All measurements on the box were made 400 metres from the APRF. This is approximately two neutron mean free paths, and is sufficient for the neutron and gamma-ray spectra to reach equilibrium shapes; that is, the shapes of the spectra do not change with increasing distance beyond this range. The 400 metre position has been designated by Allied Engineering Publication 14 (ref 3) as the primary reference location for performing fission-weapon shielding measurements.

The neutron kerma at 400 metres is approximately 3.3 mR/kWhr and the gamma kerma approximately 1.3 mR/kWhr, for a neutron-to-gamma ratio of 2.5.

7. The Protection Factor

The shielding effectiveness of the various configurations considered in this report is quantified as protection factors. The shielding against the total free-field environment is determined by the total protection factor (TPF), defined as follows:

$$\mathrm{TPF} = \frac{\text{free field dose}}{\text{total vehicular dose}} \tag{1}$$

Since neutrons and secondary gamma rays are important biological considerations, a more complete shielding analysis is accomplished by further defining a neutron protection factor (NPF) and gamma protection factor (GPF):

$$\mathrm{NPF} = \frac{\text{free field neutron dose}}{\text{direct neutron and vehicular } n,\gamma \text{ dose}} \tag{2}$$

$$GPF = \frac{\text{free field gamma dose}}{\text{gamma dose due to all extra-vehicular sources}} \quad (3)$$

The neutron protection factor quantifies the shielding of free-field neutrons. Similarly, the gamma protection factor quantifies the shielding of gamma radiation resulting from n,γ reactions in the air and ground. The prompt, or fission, gamma radiation from the source can be ignored relative to this induced gamma radiation. The neutron and vehicular n,γ dose usually accounts for about 85 percent of the total vehicular dose.

8. The Reduction Factor

In determining the effectiveness of a radiation shield, the neutron protection factor is usually the most important criterion. In a reactor experiment, the NPF can be estimated but cannot be measured directly as it is impossible to distinguish between gamma-rays resulting from interactions within a vehicle and those transmitted from outside the vehicle. Therefore, the data from the reactor measurements are reported in terms of the neutron reduction factor (NRF) and the gamma reduction factor (GRF), defined as follows.

$$NRF = \frac{\text{free field neutron dose}}{\text{vehicular neutron dose}} \quad (4)$$

$$GRF = \frac{\text{free field gamma dose}}{\text{vehicular gamma dose}} \quad (5)$$

VCS results can be presented as reduction factors or protection factors.

II. PROCEDURES

1. Target Description

The target selected for comparing the various codes was simple, but adequate for demonstrating the secondary gamma production problem with VCS. It was a simple cubic steel box, two feet on each edge. The walls of the box were two inches thick. Comparisons were make for four radiation liner configurations: two inches polyethylene (P.) inside, one inch P. inside/ one inch P. outside, two inches borated polyethylene (B.P.) inside, and one inch B.P inside /one inch B.P. outside. The one inch P./one inch P. outside configuration is the most dramatic example of the inadequacy of the secondary gamma treatment in VCS. The dose is calculated for the center of the box in each case.

III. RESULTS

Table 3 contains comparisons between VCS, VCS/69, and MCNP calculations; and APRF measurements for four simple box configurations.

Table 3 -- Box Results

Configuration	Source	TPF	NRF	GRF
Poly 1"in/1"out	APRD	2.7	6.2	1.2
	VCS	4.5	6.5	2.8
	MCNP	2.9	6.0	.93
	VCS/69	3.1	7.1	1.4
Poly 2" Inside	APRD	2.2	5.6	.94
	VCS	2.9	7.0	1.3
	MCNP	2.4	5.9	.71
	VCS/69	2.5	7.4	.80
B.Poly 1"in/1"out	APRD	4.2	4.6	3.4
	VCS	4.8	6.0	3.3
	MCNP	4.6	5.7	2.6
	VCS/69	4.9	6.8	2.6
B.Poly. 2" Inside	APRD	3.1	5.3	1.7
	VCS	3.7	6.2	2.0
	MCNP	3.5	6.8	1.1
	VCS/69	3.7	6.8	1.5

IV. DISCUSSION

Both MCNP and VCS/69 were successful in eliminating the secondary gamma problem found with VCS, as best shown by the GRF values for the 1"in/1"out polyethylene configuration. MCNP gives TPF values which agree a little better with measurements than does VCS/69. NRF values are usually better for MCNP as well, but the neutron dose is almost negligible for these examples.

V. CONCLUSIONS

VCS/69 results were noticeably better than VCS results in one out of the four configurations, one in which the secondary gamma production in steel was a major contributor to the interior dose. Because the accurate description of secondary gamma production in steel is important in a large number of armored vehicle shielding problems, VCS/69 is a significant improvement over VCS. Although MCNP has several advantages over VCS/69, the protection factors that were calculated were not noticeably better. Since considerable work would be involved in devising a transport methodology based on MCNP suitable for armored vehicles, present evidence does not warrant making such a change.

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